NMC
Committed to Nuclear Excellence

Prairie Island Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

October 29, 2004

L-PI-04-109
10 CFR 50.54(f)

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
11555 Rockville Pike
Rockville, Maryland 20852

Prairie Island Nuclear Generating Plant Units 1 and 2
Dockets 50-282 and 50-306
License Nos. DPR-42 and DPR-60

60-Day Response to Generic Letter 2004-01, "Requirements for Steam Generator Tube Inspections"

On August 30, 2004, the Nuclear Regulatory Commission (NRC) transmitted Generic Letter (GL) 2004-01. Enclosure 1 contains the Nuclear Management Company, LLC (NMC) 60-day response to GL 2004-01 for the Prairie Island Nuclear Generating Plant.

Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

I declare under penalty of perjury that the foregoing is true and correct. Executed on October 29, 2004.

Joseph M. Solymossy
Site Vice President, Prairie Island Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosure (1)

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC

A115

ENCLOSURE 1
GENERIC LETTER 2004-01
PRAIRIE ISLAND NUCLEAR GENERATING PLANT 60-DAY RESPONSE

Nuclear Regulatory Commission (NRC) Requested Information

1) Addressees should provide a description of the SG tube inspections performed at their plant during the last inspection. In addition, if they are not using SG tube inspection methods whose capabilities are consistent with the NRC's position, addressees should provide an assessment of how the tube inspections performed at their plant meet the inspection requirements of the TS in conjunction with Criteria IX and XI of 10 CFR Part 50, Appendix B, and corrective action taken in accordance with Appendix B, Criterion XVI. This assessment should also address whether the tube inspection practices are capable of detecting flaws of any type that may potentially be present along the length of the tube required to be inspected and that may exceed the applicable tube repair criteria.

Nuclear Management Company, LLC (NMC) Response

Prairie Island, Unit 1 (PI-1)

The last inspection on PI-1 was conducted during the 1R22 refueling outage in November 2002[1]. The inspection scope is provided in Table 1 below:

TABLE 1 Number and Extent of Tubes Inspected				
SCOPE	**EXTENT④**	**PROBE TYPE**	**S/G 11**	**S/G 12**
Full Length①	TEC-TEH	Bobbin	100%	100%
Rows 1 and 2 U-Bends	07C-07H	MRPC	100%	100%
Hot Leg Tubesheets	TEH-TSH+3"	MRPC	100%	100%
Sleeves Full Length	TEH-STH	MRPC	N/A	25%
Sleeves Partial Length	TSH-STH	MRPC	N/A	75%②
Hot Leg Roll Plugs	UPH-LPH	MRPC	25%	25%
Post In Situ Pressure Test	Various	MRPC	100%	100%
Supplemental③	Various	MRPC	100%	100%

[1] Prairie Island Unit 1 steam generators are being replaced during the Fall 2004 (1R23) refueling outage.

TABLE 1 Number and Extent of Tubes Inspected				
SCOPE Free Span Dents	EXTENT④ Various	PROBE TYPE MRPC	S/G 11 25%	S/G 12 25%
Plug Visual	N/A	N/A	100%	100%
Sleeve Visual	N/A	N/A	100%	100%

① Except the bend portion of rows 1 and 2 u-bends and the sleeved portion of sleeved tubes.
② One sleeved tube (R4C76) was not inspectable due to an obstruction and was plugged.
③ ADR, CUD, DEP, DNI, DNT ≥ 5.0 Volts at a tube support plate, DNT ≥ 5.0 Volts at the top of the tube sheet, DRI, DSI, DTI, INR ≥ 1.5 Volts at a tube support plate, MBM, MRI, NQI, PLP (Bound MRPC PLP's), PSI, Cold Leg Thinning ≥ 40% or < 40% and ≥ 1.5 Volts.
④ See Attachment 1 for landmark locations used for examination extents.

Acronyms/Definitions:
MRPC = Motorized Rotating Pancake Coil
ADR = Absolute Drift
CUD = Copper Deposit
DEP = Deposit
DNI = Dent with Indication
DNT = Dent
DRI = Distorted Roll Transition with Indication
DSI = Distorted Support Signal with Indication
DTI = Distorted Tube Sheet Signal with Indication
INR = Indication Not Reportable
MBM = Manufacturing Burnish Mark
MRI = Mix Residual Indication
NQI = Non-Quantifiable Indication
PLP = Possible Loose Parts
PSI = Possible Support Indication
S/G = Steam Generator

The inspection techniques employed at PI-1 are consistent with the NRC's position in that all known and potential damage mechanisms are sampled between 25 and 100 percent with techniques qualified in accordance with the Electric Power Research Institute (EPRI) Pressurized Water Reactor Steam Generator Examination Guidelines, Appendix H.

The inspection techniques employed at PI-1 are capable of detecting all potential damage mechanisms (volumetric, axial and circumferential) at the time of the inspection in regions of the tubes where they are known or postulated to occur as specified in the Degradation Assessment that may exceed the applicable repair criteria.

The detection capability of each technique employed for PI-1 is defined in the applicable EPRI or Vendor Examination Technique Specification Sheet and is accounted for in both the Condition Monitoring and Operational Assessment.

Prairie Island, Unit 2 (PI-2)

The last inspection on PI-2 was conducted during the 2R22 refueling outage in September 2003. The inspection scope is provided in Table 2 below:

TABLE 2 Number and Extent of Tubes Inspected				
SCOPE	EXTENT③	PROBE TYPE	S/G 21	S/G 22
Full Length①	TEC-TEH	Bobbin	100%	100%
Rows 1 through 11 U-Bends	07C-07H	MRPC	100%	100%
Hot Leg Tubesheets	TEH-TSH+3-6"④	MRPC	100%	100%
Cold Leg Tubesheets	TEC-TSC+1"	MRPC	20%	20%
Hot Leg Roll Plugs	UPH-LPH	MRPC	25%	25%
Post In Situ Pressure Test	Various	MRPC	N/A	100%
Supplemental②	Various	MRPC	100%	100%
Free Span Dents ≥ 5.0 Volts	Various	MRPC	25%	25%
Plug Visual	N/A	N/A	100%	100%
Baseline new Re-Rolls	TEH-TSH	Bobbin/MRPC	100%	100%

① Except the bend portion of rows 1 through 4 u-bends.
② ADR, CUD, DEP, DNI, DNT ≥ 5.0 Volts at a tube support plate, DNT ≥ 5.0 Volts at the top of the tube sheet, DRI, DSI, DTI, INR ≥ 1.5 Volts at a tube support plate, MBM, MRI, NQI, PLP (Bound MRPC PLP's), PSI, Cold Leg Thinning ≥ 40% or < 40% and ≥ 1.5 Volts.
③ See Attachment 1 for landmark locations used for examination extents.
④ TSH +3" except sludge pile region (inclusive of rows 12 through 22 and column 28 through 50) which will be examined TSH +6".

The inspection techniques employed at PI-2 are consistent with the NRC's position in that all known and potential damage mechanisms are sampled between 20 and 100 percent with techniques qualified in accordance with EPRI Pressurized Water Reactor Steam Generator Examination Guidelines, Appendix H.

The inspection techniques employed at PI-2 are capable of detecting all potential damage mechanisms (volumetric, axial and circumferential) at the time of the inspection in regions of the tubes where they are known or postulated to occur as specified in the Degradation Assessment that may exceed the applicable repair criteria.

The detection capability of each technique employed for PI-2 is defined in the applicable EPRI or Vendor Examination Technique Specification Sheet and is accounted for in both the Condition Monitoring and Operational Assessment.

NRC Requested Information

2) *If addressees conclude that full compliance with the TS in conjunction with Criteria IX, XI and XVI of 10 CFR Part 50, Appendix B, requires corrective actions, they should discuss their proposed corrective actions (e.g., changing inspection practices consistent with the NRC's position or submitting a TS amendment request with the associated safety basis for*

limiting the inspections) to achieve full compliance. If addressees choose to change their TS, the staff has included in the attachment suggested changes to the TS definitions for a tube inspection and for plugging limits to show what may be acceptable to the staff in cases where the tubes are expanded for the full depth of the tubesheet and where the extent of the inspection in the tubesheet region is limited.

NMC Response

The S/G tube inspections at Prairie Island are in full compliance with the NRC's position in regard to Technical Specifications (TS) in conjunction with Criteria IX, XI and XVI of 10 CFR Part 50, Appendix B. Therefore, this item is not applicable and no corrective actions are required.

NRC Requested Information

3) *For plants where SG tube inspections have not been or are not being performed consistent with the NRC's position on the requirements in the TS in conjunction with Criteria IX, XI, and XVI of 10 CFR Part 50, Appendix B, the licensee should submit a safety assessment (i.e., a justification for continued operation based on maintaining tube structural and leakage integrity) that addresses any differences between the licensee's inspection practices and those called for by the NRC's position. Safety assessments should be submitted for all areas of the tube required to be inspected by the TS where flaws have the potential to exist and inspection techniques capable of detecting these flaws are not being used, and should include the basis for not employing such inspection techniques. The assessment should include an evaluation of (1) whether the inspection practices rely on an acceptance standard (e.g., cracks located at least a minimum distance of x below the top of the tube sheet, even if these cracks cause complete severance of the tube) which is different from the TS acceptance standards (i.e., the tube plugging limits or repair criteria), and (2) whether the safety assessment constitutes a change to the "method of evaluation" (as defined in 10 CFR 50.59) for establishing the structural and leakage integrity of the joint. If the safety assessment constitutes a change to the method of evaluation under 10 CFR 50.59, the licensee should determine whether a license amendment is necessary pursuant to that regulation.*

NMC Response

The S/G tube inspections at Prairie Island are consistent with the NRC's position on the requirements in the TS in conjunction with Criteria IX, XI, and XVI of 10 CFR Part 50, Appendix B. Therefore, this item is not applicable and no safety assessment is required.

Attachment 1

LANDMARK	LOCATION	DESCRIPTION
TEH	0.01	Tube End Hot leg
LPH	0.75	Lower plug roll transition
UPH	2.00	Upper plug roll transition
TRH	2.75	original Tube Roll Hot leg
RTR	3.00	original Roll Transition Reroll
1BH	3.75	1st additional reroll Bottom roll transition Hot leg
1TH	5.00	1st additional reroll Top roll transition Hot leg
2BH	5.75	2nd additional reroll Bottom roll transition Hot leg
1HH	6.00	1st additional reroll top of Hydraulic transition Hot leg
2TH	7.00	2nd additional reroll Top roll transition Hot leg
2HH	8.00	2nd additional reroll top of Hydraulic transition Hot leg
EBH	16.38	Elevated reroll Bottom roll transition Hot leg
ETH	18.38	Elevated reroll Top roll transition Hot leg
TSH	21.40	Tube Sheet Hot leg
BUH	25.00	sleeve Bottom of Upper Hydraulic transition hot leg
WCH	25.50	sleeve Weld Centerline Hot leg
TUH	26.00	sleeve Top of Upper Hydraulic transition hot leg
STH	27.00	Sleeve Top Hot leg
01H	71.53	1st tube support plate Hot leg
02H	122.03	2nd tube support plate Hot leg
03H	172.53	3rd tube support plate Hot leg
04H	223.03	4th tube support plate Hot leg
05H	273.53	5th tube support plate Hot leg
06H	324.03	6th tube support plate Hot leg
07H	374.53	7th tube support plate Hot leg
NV1	VARIES	New anti Vibration 1st
NV2	VARIES	New anti Vibration 2nd
NV3	VARIES	New anti Vibration 3rd
NV4	VARIES	New anti Vibration 4th
07C	-374.53	7th tube support plate Cold leg
06C	-324.03	6th tube support plate Cold leg
05C	-273.53	5th tube support plate Cold leg
04C	-223.03	4th tube support plate Cold leg
03C	-172.53	3rd tube support plate Cold leg
02C	-122.03	2nd tube support plate Cold leg
01C	-71.53	1st tube support plate Cold leg
TSC	-21.40	Tube Sheet Cold leg
TRC	-2.75	original Tube Roll Cold leg
UPC	-2.00	Upper Plug roll expansion Cold leg
LPC	-0.75	Lower Plug roll expansion Cold leg
TEC	-0.01	Tube End Cold leg